04-423

IMPLEMENTING AGREEMENT
BETWEEN
THE NUCLEAR REGULATORY COMMISSION
OF THE UNITED STATES OF AMERICA
AND
THE CONSEJO DE SEGURIDAD NUCLEAR OF THE KINGDOM OF SPAIN
IN
THE AREA OF NUCLEAR SAFETY RESEARCH

The United States Nuclear Regulatory Commission, hereinafter referred to as USNRC, and the Consejo de Seguridad Nuclear of the Kingdom of Spain hereinafter referred to as CSN;

Considering that the USNRC and the CSN, hereinafter referred to as the Parties:

1. Have a mutual interest in cooperation in the field of nuclear safety research with the objective of improving and thus ensuring the safety of civilian nuclear installations on an international basis;

2. Recognize a need to equitably share the resources, the outcomes resulting from this research and the effort required to develop those resources; and

3. Are Parties to the five-year "Arrangement between the USNRC and the CSN for the Exchange of Technical Information and Cooperation in Nuclear Safety Matters," signed on September 19, 2000, hereinafter referred to as the "Arrangement," and that this safety research cooperation is being undertaken as an implementation of this Arrangement;

4. Have been cooperating since September 20, 1996, under an existing five year Implementing Agreement Between the United States Nuclear Regulatory Commission and the Consejo de Seguridad Nuclear of the Kingdom of Spain (CSN) in the Area of Nuclear Safety Research;

Have agreed as follows:

ARTICLE I - PROGRAM COOPERATION

The Parties, in accordance with the provisions of this Implementing Agreement and subject to applicable laws and regulations in force in their respective Countries, will undertake a program for cooperative research in nuclear safety programs sponsored by the USNRC as well as those sponsored by the CSN. This five-year cooperative program will include:

a) thermal-hydraulic computer code development and assessment studies (see Technical Appendix);
b) technical information exchange in the areas of reliability and risk, accident management and human and organizational factors, and advanced instrumentation and control;
c) research topics concerning aging of nuclear plant components and materials research including storage and transportation of spent nuclear fuel;

d) participation by the CSN in the USNRC Cooperative Severe Accident Research Program including high burnup fuel behavior, as well as participation in the USNRC Cooperative Probabilistic Risk Assessment Program.

e) research on radionuclide transport in the environment and high activity waste management;

f) research topics concerning site characterization issues; and

g) participation by CSN in other nuclear safety research areas to be mutually determined while this Implementing Agreement is in force.

ARTICLE II - FORMS OF COOPERATION

Cooperation between the Parties may take the following forms:

A. Exchange of information in the form of technical reports, experimental data, computational codes, correspondence, newsletters, visits, joint meetings, and such other means as the Parties agree.

B. Temporary assignment of personnel of one Party or of its contractors to the laboratory or facilities owned by the other Party or in which it sponsors research; each such assignment to be considered on a case-by-case basis and be the subject of a separate attachment-of-staff arrangement between appropriate representatives of the recipient and assigning organizations.

C. Execution of joint programs and projects, including those involving a division of activities between the Parties; each such joint program and project shall be considered on a case-by-case basis and may be the subject of a separate agreement between the Parties, as appropriate.

D. Use by one Party of facilities that are owned by the other Party or in which research is being sponsored by the other Party; such use of facilities shall be the subject of separate agreements between the relevant entities and may be subject to commercial terms and conditions.

E. If either Party wishes to visit, assign personnel, or use the facilities owned or operated by entities other than the Parties to this Implementing Agreement, the Parties recognize that prior approval of such entities will in general be required regarding terms upon which such visit, assignment, or use shall be made.

F. Any other form agreed between the Parties.

ARTICLE III - SCOPE OF IMPLEMENTING AGREEMENT

Subject to the availability of appropriated funds, the USNRC and the CSN will cooperate in the areas of nuclear safety research outlined in Article I. The specific details of this cooperation are outlined in the Technical Appendix which is an integral part of this Implementing Agreement.

ARTICLE IV - ADMINISTRATION OF THE IMPLEMENTING AGREEMENT

A. The USNRC and the CSN will each designate one representative to coordinate and determine the detailed implementation of this follow-on Implementing Agreement. These representatives may, at their discretion, delegate this responsibility to the appropriate technical staff with respect to a given issue. The single designated representative will be referred to as an Administrator of this Implementing Agreement.

B. Information on matters related to organization, budget, personnel, or management may be restricted under this Implementing Agreement.

C. The USNRC and the CSN will endeavor to select technical personnel for assignment to these cooperative programs who can contribute positively to the programs. USNRC and Spanish technical personnel assigned for extended periods will be considered visiting scientists (non-salaried) within the programs in this Implementing Agreement and will be expected to participate in the conduct of the analyses and/or experiments as necessary.

D. Each Party to this Implementing Agreement will have access to all reports written by its partner's technical personnel assigned to the respective programs that derive from its participation in those programs.

E. Travel costs, living expenses, and salaries will be borne by the Parties who incurred them unless specified otherwise.

ARTICLE V - EXCHANGE AND USE OF INFORMATION AND INTELLECTUAL PROPERTY

A. General

The Parties support the widest possible dissemination of information provided or exchanged under this Implementing Agreement, subject both to the need to protect proprietary or other confidential or privileged information as may be exchanged hereunder, and to the provisions of Article III and the Intellectual Property Addendum of the Arrangement between the USNRC and the CSN signed on September 20, 2000, which shall govern this implementing agreement.

B. Other Considerations.

1. Nothing contained in this Agreement will preclude a Party from using or disseminating information received without restriction by a Party from sources outside of this Agreement.

2. All USNRC computer codes disseminated under this Agreement are to be considered privileged information unless otherwise noted, are protected as such by the USNRC, and shall be treated likewise by CSN. They are, in particular, subject to all the provisions of this Article including the requirements for an agreement of confidentiality (Article V) prior to dissemination, with the exception that they need not be marked with the restrictive designation. The codes are

subject to this protection in both object and source forms and as recorded in any media.

3. The USNRC codes and other related analytical techniques covered under this Agreement and any improvements, modifications or updates to such codes or techniques, are for the purpose of reactor and plant systems safety research and licensing and will not be used for commercial purposes, or for other benefits not related to the study of reactor safety without the prior consent of USNRC.

   Among the code uses that will be permitted under this Agreement are those related to research in the reactor safety area and analyses performed by the members or their contractors that can assist regulators and plant personnel in assessing the safety of the plant, analyzing operating events, and training of operators. Specific examples of permitted analyses include: design basis accidents (e.g., loss-of-coolant-accidents), anticipated transients, accident management and emergency operating procedures, mid-loop operation, analyses to support PRA success criteria, power upgrades and reload.

   Prohibited uses of the code include: (1) *analyses to develop a new* reactor design and (2) analyses to support power upgrades and reload in the U.S. unless performed by a U.S. subsidiary.

4. The USNRC codes and other related analytical techniques will not be advertised directly or by implication to obtain contracts related to the construction or servicing of nuclear facilities, nor will advertising imply that the USNRC has endorsed any particular analyses or techniques.

5. All reports published within the scope of this Agreement and all meetings held will be in English.

ARTICLE VI - FINANCIAL CONSIDERATIONS

In addition to the technical contributions indicated under Section B of the Technical Appendix, the CSN will contribute financially to the USNRC programs included in this Implementing Agreement. These financial considerations are for only three years to allow the CSN the option to extend and/or modify the specific technical scope in the cooperative programs requiring CSN cash contributions during the last two years of this Implementing Agreement.

In addition to the in-kind contributions identified in Part B of the Technical Appendix, the CSN will contribute $100,000 USD per year to the Cooperative Nuclear Safety Research Program. Payment will be due on the date that this implementing Agreement is signed, and on the anniversary of that date for each subsequent calendar year. The funds will be distributed by the USNRC Office of Nuclear Regulatory Research in the following manner: Probabilistic Risk Assessment Research ($15K), Human Performance Research and Advanced Instrumentation and Control Systems Research ($15K), Severe Accident Research and High Burnup Fuel Research ($55K), Aging and Component Integrity Research ($15K). The Thermal Hydraulic Code Applications and Maintenance Program (CAMP) cash contribution requirements are to be determined based on the number of participants and the number of computer codes to be included in this program. The CSN contributes its share of the Spanish contribution to the CAMP Program under a separate agreement.

ARTICLE VII - DISPUTES AND WARRANTY OF INFORMATION

A. All costs arising from implementation of this Implementing Agreement shall be borne by the Party that incurs them except when specifically agreed to otherwise by both Parties.

B. Cooperation under this Implementing Agreement shall be in accordance with the laws and regulations of the respective countries. Any dispute or questions between the Parties concerning the interpretation or application of this Implementing Agreement arising during its term shall be settled by mutual agreement of the Parties.

C. Information furnished by one Party to the other under this Implementing Agreement shall be accurate to the best knowledge and belief of the Party supplying the information. However, neither Party gives any warranty as to the accuracy of such information or shall have any responsibility for the consequences of any use to which such information may be put by the other Party or by any third Party.

ARTICLE VIII - FINAL PROVISIONS

A. This Implementing Agreement shall enter into force upon signature, with effect from September 21, 2001, and shall remain in force for a period of five years, contingent upon the successful renewal of the Arrangement. All information protected by provisions of this Implementing Agreement as proprietary, confidential, privileged, or otherwise subject to restriction on disclosure shall remain so protected indefinitely, unless mutually agreed to in writing.

B. Either Party may withdraw from the present Implementing Agreement after providing the other Party written notice at least 180 days prior to its intended date of withdrawal. The Party not withdrawing shall reserve the right to determine if the withdrawal will result in the other Party receiving a disproportionate share of the expected benefit from this Implementing Agreement. If so, both Parties will endeavor to reach an equitable settlement of the matter through negotiation.

C. The Parties to this Implementing Agreement reserve the right to modify or extend the specific activities described in the Technical Appendix within the intended scope of the Implementing Agreement upon written concurrence of their Administrators.

D. The USNRC and the CSN recognize the benefits of international cooperation and will endeavor to obtain a mutually agreeable continuation of this Implementing Agreement before its expiration.

DONE in duplicate in the English and Spanish languages, both texts being equally authentic.

FOR THE UNITED STATES NUCLEAR REGULATORY COMMISSION:

Name: William D. Travers /for
William D. Travers

Title: Executive Director for Operations

Date: 9/3/02

Place: Rockville, MD 20852

FOR THE CONSEJO DE SEGURIDAD NUCLEAR OF THE KINGDOM OF SPAIN:

Name:
Antonio Morales

Title: General Secretary

Date: 9/3/02

Place: Madrid, Spain CSN Headquarters

CERTIFIED A TRUE COPY
BY Emile L. Julian
Office of the Secretary

TECHNICAL APPENDIX

NUCLEAR SAFETY RESEARCH PROGRAMS

Section A. USNRC Program Scope

1. Thermal-Hydraulic Research and Code Assessment

Coordination and Program Management. The Thermal-Hydraulic Code Applications and Maintenance Program (CAMP), a follow-up program to the International Code Assessment Program (ICAP), is coordinated by the USNRC. This program is fully described under a separate agreement between the USNRC and the CSN and is noted here only to acknowledge its existence and relationship to other nuclear safety programs described below.

2. Probabilistic Risk Analysis (PRA)

The international cooperative research effort in probabilistic risk assessment (PRA), has been divided into four general areas of research: (1) Methods Development, (2) Analysis of Operating Events, (3) Development of Advanced PC-Based PRA Software, and (4) Regulatory Applications of PRA. The activities planned in each of these areas are broadly described in the following sections.

1. Methods Development

It is generally recognized that the broad application of PRA to support regulatory decision-making requires methods improvements in a number of risk-significant areas. Among the areas needing improvement are treatment of fire risk, equipment aging, human reliability, and the reliability and risk impact of digital control systems. NRC programs in these areas are as follows:

a. Fire Risk

The overall purpose of the fire risk research program is to provide technical information in support of the NRC's Risk-Informed Regulation Implementation Plan (RIRIP). In particular, the program will develop fire PRA methods, tools, data, results, and insights needed by the agency to perform risk-informed decision making.

The fire risk program includes activities that: 1) improve qualitative and quantitative understanding of the risk contribution due to fires in operating nuclear power plants (NPPs) and other facilities regulated by the NRC; 2) support ongoing or anticipated fire protection activities in the NRC program offices, including the development of risk-informed, performance-based approaches to fire protection for operating NPPs; and 3) evaluate current fire PRA methods and tools and develop improved tools (as needed to support the preceding objectives).

Previous work has led to: the development of improved methods, tools, and data in a number of areas, including circuit analysis, fire detection and suppression analysis, and uncertainty analysis; and to the development of fire PRA insights from reviews of past significant fire events. Ongoing work, identified in a draft research plan prepared in 2001, includes efforts to: apply the improved methods, tools, and data in plant-specific studies; develop (in cooperation with a number of international organizations) an improved understanding of the uncertainties and limitations in current fire models; and support ongoing fire-related regulatory efforts (e.g., the NRC's fire protection Significance Determination Process).

b. Equipment Aging

The objective of this research effort is to assess the feasibility of using expert judgment, reliability-physics based methods, and the results of the Nuclear Plant Aging Research Program (NPARP) to incorporate the effects of aging into an integrated probabilistic risk assessment. This assessment will be done through trial applications of an existing PRA using the SAPHIRE risk assessment computer model. Ultimately, measures of core damage frequency and associated uncertainties will be obtained for different periods of time during plant life. As necessary, SAPHIRE models will be modified to incorporate the systems, structures, and components shown to be risk-significant.

The output from this program will be the selection of a plant model (systems, structures and components) and relevant aging mechanisms for study. The methods described above will be applied to develop time-dependent unavailability predictions for those structures, systems, and components selected earlier for study. The current phase of this work was completed in late 1999 and has been published. Based on the results of this work, additional research has been initiated to model the aging of in-containment electrical cables in a PRA .

c. Human Reliability

The general objectives of the human reliability analysis (HRA) research are to: 1) develop improved human reliability analysis (HRA) methods, tools (including guidance), and data needed to support NRC regulatory activities, including the broad implementation of risk-informed regulation; 2) Develop HRA insights to support the development of technical bases for addressing identified or potential safety issues; and 3) provide HRA support for the planning and execution of NRC programs and activities outside the immediate scope of the RIRIP.

Previous work has led to the development of ATHEANA, an improved method for HRA that focuses on the identification of error forcing contexts that increase the likelihood of human errors, and the demonstration application of that method at the Seabrook NPP. Ongoing work, identified in a research plan prepared in 2001, includes: the application of ATHEANA in the assessment of pressurized thermal shock (PTS) risk in support of efforts to re-examine the technical basis for 10 CFR 50.61, the PTS rule; the development of an improved method for HRA quantification that includes the use of evidence from a variety of sources; the identification and evaluation of potentially useful sources of HRA data; and the development of HRA guidance for various users (including reviewers).

d. Digital Systems Reliability and Risk

The increased use of digital instrumentation and control systems in nuclear power plants is introducing some unique reliability and risk issues. This project will be focused on providing methods for more quantitative, probabilistic assessments of digital systems reliability and their impact on overall plant risk, including hardware and software reliability and human-system interface issues. A comprehensive research program plan is being prepared and is expected to be available in mid 2002.

2. Analysis of Operating Events

a. ASP Program

The Accident Sequence Precursor (ASP) Program was established by the NRC in 1979 in response to the Risk Assessment Review Group report (see NUREG/CR-0400, September 1978). The primary objective of the ASP Program is to systematically evaluate U.S. nuclear plant operating experience to identify, document, and rank operating events most likely to lead to inadequate core cooling and severe core damage (precursors), if additional failures had occurred.

The secondary objectives of the ASP Program are

- To categorize the precursors by their plant-specific and generic implications,

- To provide a measure for trending nuclear plant core damage risk, and
- To provide a partial check on probabilistic risk assessment (PRA)-predicted dominant core damage scenarios.

The program is also used to monitor the agency's performance against the following NRC performance goals:

No more than one event per year identified as a significant precursor (i.e., conditional core damage probability or importance > 1 x 10-3) of a nuclear reactor accident.

No statistically significant adverse industry trends in safety performance.

Events and conditions from licensee event reports, inspection reports, and special requests from NRC staff are reviewed for potential precursors. These potential precursors are analyzed, and a conditional core damage probability (CCDP) is calculated by mapping failures observed during the event onto accident sequences in risk models. An event with a CCDP or a condition with a change in core damage probability greater than or equal to 1.0 x 10-6 is considered a precursor in the ASP Program.

b. SPAR Model Development Program

The Standardized Plant Analysis Risk (SPAR) models are the analysis tool used by staff analysts in many regulatory activities, including the ASP Program. The SPAR models have evolved from two sets of simplified event trees that were used initially to perform precursor analyses in the early 1980s. One set of event trees was used for boiling-water reactors (BWRs) and one set for pressurized-water reactors (PWRs) to model plant response to the same set of initiating events. Event trees are linked to fault trees that are used in the quantification of event tree sequences.

The Level 1 SPAR models are developed for all U.S. nuclear power plants. Currently, Revision 2 of these models contain four initiating events (transients, loss of offsite power, small loss-of-coolant accidents (LOCA), and steam generator tube rupture (PWRs)) and associated fault trees. Anticipated transients without scram and station blackout events are modeled as transfers from the transient and loss of offsite power events trees. Revision 3 of the SPAR models is being developed. These models include six additional initiating events (medium LOCA, large LOCA, loss of vital dc bus, loss of raw cooling water, loss of component cooling water (PWRs), and interfacing system LOCA). Revision 3 models include improved human error modeling, component-level recovery modeling, and uncertainty analysis capability. An automated user interface (the GEM program) is provided for performing the types of plant evaluation most frequently encountered during the analysis of ASP events. Production of the Revision 3 models will be completed in 2002; on-site certifications of all models are planned for completion by 2004.

In addition to the Level 1 full power operation models, generic models for low-power and shutdown operations, and Level 2 large early release frequency (LERF) analysis are being developed for several plant categories. Instructions for adapting these generic models for plant-specific applications will be included with the models. These models will be completed by 2005. In addition to these efforts, in 2003, work will be started on the development of tools for analyzing events and conditions associated with external event initiators.

3. Development of PC-Based PRA Software

The NRC has developed and maintains the SAPHIRE (Systems Analysis Programs for Hands-on Analysis Integrated Reliability Evaluations) PRA computer code. SAPHIRE offers a state-of-the-art capability for assessing the risk associated with any complex system or facility. In particular SAPHIRE can be used to assess the risk associated with nuclear power plants in terms of core damage frequency (Level 1 PRA) and containment performance and radioactive releases (Level 2 PRA). SAPHIRE includes GEM, a separate subroutine that provides a simplified user interface for performing analysis using SPAR models, discussed above.

Both the continual advancement of the state-of-the-art in the use of computers and the continual expansion of the use of risk-information in the NRC's decision-making, necessitate continual maintenance and improvement of SAPHIRE.

It is expected that, in FY 2002-2003, this program will continue to provide software maintenance and user support and expand SAPHIRE capabilities by: decreasing size limitations (on the number of basic events, fault trees, sequences, end states, etc. handled by SAPHIRE), speeding up cutset generation and data analysis using multiple processors, adding work group project integration capability, and creating a web-page type user interface with the goal of reducing complexity without losing SAPHIRE's functionality. Furthermore, SAPHIRE's documentation will be revised by issuing a new report for the Windows Versions 6 and 7.

4. Regulatory Applications of PRA

a. Changes to Reactor Regulations

NRC has been actively pursuing the increased use of PRA methods, models, and insights to support regulatory decisions. Among the active programs are those which use PRA results to identify changes needed in reactor safety requirements. There are currently two regulations (10 CFR 50.44 and 10 CFR 50.46) that the staff is revising based on current risk information and research results. Proposed rules for these changes are expected in 2002 and 2003, respectively. In parallel, the staff continues its study of other possible rule changes.

b. Regulatory Guidance on PRA

The NRC staff is also developing a new regulatory guide (RG) that will provide guidance to licensees on how to use PRA standards and industry peer review programs to demonstrate that the risk metric input to a risk-informed decision is technically defensible. Accompanying this new RG will also be a Standard Review Plan (SRP) chapter. The main body of the RG will provide guidance on the use of PRA standards and industry guidance by licensees to determine the level of confidence that can be afforded PSA insights/results in support of decision-making. The staff's endorsement of the standards and industry program will be the appendices to this RG. Release of the RG for public comment is expected in August 2002.

c. Risk of Dry Cask Fuel Storage

NRC is performing a pilot PSA of a spent fuel dry cask storage system, the Holtec International HI-STORM 100. This cask is being studied at a specific BWR site where the operations can be observed and modeled. (Although developed for a specific cask at a specific site, the analytical models developed for this preliminary study can be modified and applied to other dry cask systems at other reactor sites.) During its service life, the cask has three operational modes - handling in the reactor building, transfer to the storage pad, and storage for 20 years. In each of these modes, accidents that could result in mechanical and thermal challenges to the cask and that have the potential to cause the release of radioactive material, are postulated. Event tree/fault tree methods are used to develop logic models of plausible accident sequences. Engineering analyses are used to determine the stresses that would be imposed by the postulated events. Fracture mechanics and other engineering disciplines are used to determine the probability of a cask failing when subjected to postulated accident conditions. A human reliability analysis is used to determine the probability of accidents caused by incorrectly performed procedures, such as when the cask is moved while inside the reactor building or while being monitored during storage.

The preliminary results of the PSA suggest that the risk of the HI-STORM cask at the BWR plant is low compared to the risk of accidents involving the core of operating nuclear power plants. Events that have a high conditional probability of failing the cask have a low frequency (on the order of 10^{-6} per year or less). Conversely, events that occur with a high frequency have a low conditional probability (on the order of 10^{-6} or less) of failing the cask. Furthermore, the consequences of most of the postulated events that fracture the cask and the fuel are low because the energy driving the radionuclides from the fuel pellets is low and the inventory of radionuclides in the fuel pellets is relatively low compared to the reactor inventory. Accordingly, the risk, defined as the product of the frequency and consequences of the events, appears to be low. A draft report is expected to be completed in mid 2002.

d. Safety Goals for Nuclear Materials Regulation

After reviewing SECY 99-100, "Framework for Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards," the NRC's Commissioners directed the Office of Nuclear Materials Safety and Safeguards (NMSS) staff to implement a framework for using risk assessment in regulating the nuclear materials and waste arenas. Therefore, the NMSS staff has initiated an effort to evaluate the current regulatory process governing the NMSS licensees, conducted case studies using risk information, and developed a first draft of safety goals for nuclear materials and waste. The results from the case studies showed that safety goals for nuclear materials and waste were feasible and could be useful in risk-informing specific situations within the nuclear materials and waste arenas. Subsequently, NMSS requested that the Office of Regulatory Research (RES) assist NMSS in supporting risk-informed initiatives and activities, including continuing development of safety goals for NMSS applications.

The objective of this RES task is to further develop the nuclear materials and waste safety goals and progress toward a final version. The process of developing nuclear materials and waste safety goals is expected to be a highly collaborative effort between the RES, designated staff in the NMSS, and Brookhaven National Laboratory (BNL), the contractor for RES. The activities would include conducting international literature searches on developing safety goals in foreign countries, presenting the draft safety goals to stakeholders, soliciting comments and recommendations from stakeholders and risk experts, testing the draft safety goals, and revising draft safety goals as necessary. An updated draft safety goals report is to be expected by the end of 2002. Subsequent efforts shall be formulating an action plan to perform additional activities needed to finalize the safety goals. The additional activities would include identifying and developing subsidiary safety objectives.

3. Human Performance Research

As part of its nuclear safety research program, four current programs which may be of interest to the CSN are Deregulation, Human Reliability Analysis (HRA) Data Collection and Analysis, and Review Guidance,

Deregulation: This program was initiated as a research activity with the University of Wisconsin Center for Human Performance and Risk Analysis and includes:

- A case study report describing the impact of economic deregulation on the airline and rail industries in the U.S. and the power industry in the United Kingdom.
- A Subject Matter Expert (SME) Workshop on research that the USNRC may undertake to address issues raised in the case study report (as well as other issues identified during the workshop).
- Cooperation with Switzerland (pending) on a workshop on Deregulation and possible research on significant issues identified in the workshop

HRA DATA Collection and Analysis: This new program involves improving the quantification of human failure event probabilities and to support HRA models. This work involves:

- Defining the qualitative and quantitative data needs for HRA
- Identifying sources of archival and original data
- Developing long-term working relationships with key HRA and human factors research programs capable of generating new data.
- Collecting and analyzing data to support HRA model Development and quantification.

Review Guidance: This is a long-term research program to update guidance used in the review of control facilities at nuclear power reactors. This includes:

- Development of guidance for the review of control rooms where digital displays and controls are being introduced into existing controls control rooms.

- Development of guidance for the review of control facilities for advanced reactors including human-system interfaces, staffing, procedures, conditions of operation, etc.
- Development of guidance for the review of changes to the reliance of operator actions.

4. Advanced Instrumentation and Control System Research.

As part of the nuclear safety research program, the USNRC is developing new methods for assessing the digital instrumentation and control (I&C) systems being proposed for the deployment in nuclear power plants. The research is targeted at supporting current regulatory activities and will support future regulatory needs including advanced reactor reviews and use of probabilistic risk assessment methods in the I&C area. The current research program includes research on methods for improving the efficiency of the technical review process, developing reliability models, and developing regulatory guidance for evolving technologies. The long-term research will include developing tools to evaluate the application of new technologies to existing and advanced reactors and developing methods to incorporate digital systems' reliability information into plant PRA's. To meet the needs identified above, the USNRC is engaged in research in four general areas described below. The USNRC will provide, upon request, the published results of this program. Separate cooperative agreements in these areas may also be developed on mutual consent of both Parties.

Systems Aspects of Digital Technology: Research in this area will address both internal interactions and external factors that affect digital system performance such as electromagnetic interference and lightning.

Software Quality Assurance: Research in this area includes developing objective software quality assurance measures for use in nuclear power plant applications. The USNRC is investigation various objective criteria and software engineering techniques.

Risk Assessment of Digital I&C Systems. This research will include analysis of U.S. and international digital I&C failure data, investigate digital I&C failure data and reliability assessment methods, and quantify the risk importance of digital systems.

Emerging I&C Technology and Applications: Current plans include evaluation predictive maintenance and online monitoring systems, advanced instrumentation, smart transmitters, wireless communication and computer security.